Tennessee Valley Authority, Post Office Box 2000, Spring City, Tennessee 37381

October 26, 2012

10 CFR 50.73

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Watts Bar Nuclear Plant, Unit 1
Facility Operating License No. NPF-90
NRC Docket No. 50-390

Subject: **Licensee Event Report 390/2012-004, Automatic Reactor Trip due to Low-Low Steam Generator Level**

The enclosed Licensee Event Report provides details concerning an automatic reactor trip that occurred due to low-low steam generator level. This report is being submitted in accordance with 10 CFR 50.73(a)(2)(iv)(A), a condition that resulted in automatic actuation of the reactor protection system and automatic actuation of the auxiliary feedwater system.

There are no regulatory commitments in this letter. Please direct any questions concerning this matter to Donna Guinn, WBN Site Licensing Manager, at (423) 365-1589.

Respectfully,

D. E. Grissette
Site Vice President
Watts Bar Nuclear Plant

Enclosure
cc: See Page 2

Enclosure
cc (Enclosure):

NRC Regional Administrator - Region II

NRC Senior Resident Inspector - Watts Bar Nuclear Plant

NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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4. TITLE
Automatic Reactor Trip due to Low-Low Steam Generator Level

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									N/A	
08	28	2012	2012	- 004 -	0	10	26	2012	N/A	

9. OPERATING MODE: 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Julie Hough, Licensing Engineer	(423) 365-8048

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On August 28, 2012, at 0332 hours with Watts Bar Nuclear Plant (WBN) Unit 1 at 100 percent rated thermal power, the reactor tripped automatically on low-low steam generator (SG) #2 level. Incorrect configuration of test equipment during testing of a Pressurizer Pressure channel induced a fault into the channel, and the channel fuse and upstream circuit breaker cleared on this fault. Loss of the upstream circuit removed power from the #2 main feedwater regulating valve circuitry, causing the valve to fail closed and resulting in loss of main feedwater to SG #2. Low-low level in SG #2 resulted in an automatic reactor trip and initiation of the auxiliary feedwater system. Plant personnel immediately entered appropriate trip response procedures and stabilized the plant in Mode 3. All safety systems functioned as designed.

The root cause of this event is that procedure detail was inadequate to ensure correct configuration of the test equipment by inexperienced technicians. Procedures for circuit-intrusive activities with similar consequences were identified, placed on administrative hold, and will be updated to ensure procedural guidance meets procedure writing requirements.

This event is reported as an LER in accordance with 10 CFR 50.73(a)(2)(iv)(A) for the automatic actuation of the reactor protection system and the auxiliary feedwater system.

NARRATIVE

I. PLANT CONDITIONS:

Watts Bar Nuclear Plant (WBN) Unit 1 was operating at 100 percent rated thermal power.

II. DESCRIPTION OF EVENT:

A. Event:

On August 28, 2012, personnel were performing surveillance instruction (SI) 1-SI-68-6, 18 Month Channel Calibration Pressurizer Pressure Channel II, Loop 1-LPP-68-334. The incorrect configuration of test equipment used during performance of 1-SI-68-6 induced a fault into the channel. As a result of the incorrect configuration, the channel fuse and upstream circuit breaker cleared. Loss of the upstream circuit removed power from the #2 main feedwater regulating valve [EIIS code SJ] circuitry, causing the valve to fail closed which resulted in the loss of feed to steam generator (SG) [EIIS code AB] #2. Low-low level in SG #2 resulted in an automatic reactor trip and actuation of the auxiliary feedwater system. The unit was stabilized in Mode 3, and an event investigation was initiated. All safety systems functioned as designed.

B. Inoperable Structures, Components, or Systems that Contributed to the Event:

None.

C. Dates and Approximate Times of Major Occurrences:

Date	Time (EDT)	Event
August 28, 2012	Night shift	Instrument Maintenance personnel commenced performance of 1-SI-68-6.
August 28, 2012	03:30:49	Operations began receiving various Loop 2 steam/feedwater mismatch, low feedwater, and low SG level alarms.
August 28, 2012	03:31:32	Reactor trip due to low-low SG level.

D. Other Systems or Secondary Functions Affected:

No other systems or secondary functions were affected by this event.

E. Method of Discovery:

Control Room alarms alerted operators to the start of the event.

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NARRATIVE

II. DESCRIPTION OF EVENT (continued):

F. Operator Actions:

Following the reactor trip, the operators entered the following procedures to stabilize the plant in Hot Standby Conditions:

Emergency Operating Instruction, E-0, Reactor Trip or Safety Injection

Emergency Operating Instruction, ES-0.1, Reactor Trip Response

Abnormal Operating Instruction, AOI-17, Turbine Trip

General Operating Instruction, GO-5, Unit Shutdown from 30% Reactor Power to Hot Standby

G. Safety System Responses:

All safety systems operated as designed.

III. CAUSE OF EVENT:

A. Direct Cause:

The direct cause of this event was loss of power to control cabinet 1-Rack-19. When this rack lost power, the #2 main feedwater regulating valve failed closed.

B. Root Cause:

The root cause of this event was that the procedure detail in 1-SI-68-6 was inadequate to ensure correct configuration of test equipment (i.e., a break-in box) by inexperienced technicians. The result was a loss of power to 1-Rack-19 and subsequent reactor trip.

C. Contributing Causes:

Contributing causes were that some individuals did not “stop when unsure” when they encountered procedural guidance that was not written to their level of understanding. Additionally, risk screening processes did not identify the appropriate risk or mitigating strategies for performance of 1-SI-68-6 and management monitoring did not recognize that procedural guidance was vague for those unfamiliar with the task.

IV. ANALYSIS OF THE EVENT:

Loss of normal feedwater is analyzed in Updated Final Safety Analysis Report (UFSAR) section 15.2.8. The SG low-low level reactor protection system trip function ensures that protection is provided against a loss of heat sink to the reactor vessel.

NARRATIVE

IV. ANALYSIS OF THE EVENT (continued):

Since the plant is tripped before the SG heat transfer capability is reduced, the primary system variables maintain acceptable margin to a departure from nucleate boiling condition. In order to ensure a heat sink is restored, this condition also actuates the auxiliary feedwater system. The UFSAR loss of feedwater event analysis demonstrates that, following a loss of normal feedwater, the auxiliary feedwater system is capable of removing the stored and residual heat, thus preventing water relief from the pressurizer and subsequently a loss of water from the reactor core. In this event, all safety systems operated as designed and analyzed in the UFSAR.

V. ASSESSMENT OF SAFETY CONSEQUENCES:

Based on the above "Analysis of the Event," this event did not adversely affect the health and safety of plant personnel or the general public.

VI. CORRECTIVE ACTIONS:

A. Immediate Corrective Actions:

Work was stopped and an incident investigation was commenced. The event was entered into the corrective action program.

B. Corrective Actions to Prevent Recurrence:

Corrective actions to prevent recurrence include identifying other circuit intrusive activities with potentially similar consequences and ensuring the procedural guidance for their performance is written per procedure writing guidelines. The worst case failure for each activity will be identified in the procedure, along with necessary compensatory measures. This information will also be used to provide a reference instruction for the station for conduct of future risk screening evaluations.

VII. ADDITIONAL INFORMATION:

A. Failed Components:

None.

B. Previous LERs on Similar Events:

A review of previous reportable events for the past three years did not identify any previous reactor trip events with the same root cause.

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NARRATIVE

VII. ADDITIONAL INFORMATION (continued):

C. Additional Information:

None.

D. Safety System Functional Failure:

This event did not result in a safety system functional failure as defined by 10 CFR 50.73(a)(2)(v) and NEI 99-02.

E. Unplanned Scrams with Complications:

This event did not result in an unplanned scram with complications as defined by NEI 99-02.

VIII. COMMITMENTS:

None.